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Overview of Stellarator Divertor Studies: Final Report of LDRD Project 01-ERD-069

M.E. Fenstermacher, T.D. Rognlien, A. Koniges, M. Umansky, D.N. Hill

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M.E. Fenstermacher (PI), T.D. Rognlien, A. Koniges, M. Umansky, & D.N. Hill
Lawrence Livermore National Laboratory
Livermore, CA 94550

Abstract

A summary is given of the work carried out under the LDRD project 01-ERD-069 entitled Stellarator Divertor Studies. This project has contributed to the development of a three-dimensional edge-plasma modeling and divertor diagnostic design capabilities at LLNL. Results are demonstrated by sample calculations and diagnostic possibilities for the edge plasma of the proposed U.S. National Compact Stellarator Experiment device. Details of the work are contained in accompanying LLNL reports that have been accepted for publication.

Introduction

Modeling the physics of magnetized plasma in full three-dimensional (3D) geometry, without any axes of symmetry, is one of the grand challenges of plasma physics research. Applications range from atmospheric studies to thermonuclear fusion reactor design. At LLNL there has been an ongoing set of research projects to help develop controlled fusion energy because of its potential both as a clean, abundant energy source and as a neutron source that could transmute present radioactive material into much safer products. For magnetically confined plasma, most of the modeling to date has assumed at least one axis of symmetry for equilibrium plasma parameters. However, the National Compact Stellarator Experiment (NCSX) – to be built at the Princeton Plasma Physics Laboratory (PPPL) – is a toroidal magnetic confinement device with no axis of symmetry. The motivation for this LDRD project was to use our recognized expertise for computational simulation of the edge plasmas in axisymmetric tokamaks (2D geometry) to produce the fully 3D modeling capability that is needed for more geometrically complex geometries such as the stellarator and for other 3D magnetized plasma applications.

Purpose of the LDRD Project

The technical objectives of this LDRD project were to 1) bring our expertise to bear in the development of 3-D transport and field-line tracing codes, and (2) become strong participants in the US NCSX design activities and join the international stellarator community. Field-line tracing was our first step in analyzing the edge plasma in 3D geometry such as the stellarator. However, our long-term goal was to build-up LLNL's core competency in state-of-the-art computer modeling by collaborating with the five-person team at the Max Planck Institute for Plasma Physics (IPP) in Greifswald, Germany on the development of BORIS, a multi-species, 3-D, magnetized-plasma-fluid simulation code. This capability has given LLNL the 3-D computational techniques needed to design the plasma-interaction hardware (the divertor) for stellarators and other non-axisymmetric plasma devices The LLNL MFE group is recognized internationally for its unique capabilities in divertor physics studies combining hardware design, experimental measurements, and numerical simulations. This LDRD support has allowed us to apply our expertise to stellarator devices and placed us in excellent position to partner with PPPL on the NCSX project. As part of this work, we also have expanded our collaboration with the German stellarator program headed by Dr. Fritz Wagner and established connections to the Japanese LHD stellarator program.

Approach and Activities

The stellarator divertor studies supported by LDRD funds have focussed on two activities: 1) developing the capability to perform field-line-tracing calculations with diffusion in 3D geometry to simulate cross-field particle and energy transport in the edge region for NCSX magnetic equilibria, and 2) developing the capability to perform 3-D fluid plasma simulations of the NCSX boundary plasma. In addition we have: 1) surveyed the databases of stellarator boundary plasma measurements from existing experiments, 2) done analytical calculations of the effect of variation in field line connection length on upstream and downstream plasma parameters for NCSX that were presented at the NCSX Physics Validation Review (PVR), and 3) supplied expertise to the NCSX divertor hardware and diagnostics design presented in the NCSX Conceptual Design Review (CDR).

Technical Outcome

Field line tracing with diffusion

During this LDRD project, we developed, and implemented on LLNL computers, field-line tracing capability in 3D geometry, including a model of turbulent heat-diffusion, and applied it to the NCSX baseline design. These calculations identified the regions of the NCSX material surface where concentrated heat and particle deposition are expected and thus where protective armor must be included in the device design. These results were presented at the 2001 APS-DPP meeting and will be published in Nuclear Fusion [1]. The full report, "Magnetic Topology of a Candidate NSCX Plasma Boundary Configuration" by A. E. Koniges, et al., UCRL-JC-147845, is attached.

3D Fluid simulations with the BORIS code

We also completed the first phase of our contribution to the development of the 3D fluid simulation code BORIS. We contributed our numerical expertise in implicit solvers, interpolation methods, and fluid-neutral models to the development of BORIS. When fully developed, the BORIS simulations will give self-consistent plasma solutions from the interior edge to the plasma–material interaction surfaces. We implemented a 3D neutrals model in the BORIS package and performed multiple benchmarking tests of the code against UEDGE, our mature 2D edge-plasma transport code. The results were presented at the 15th Plasma Surface Interactions meeting and will be published in Journal of Nuclear Materials [2, 3]. The full reports, “Modeling of Localized Gas Injection in 3D Geometry,” by M.V. Umansky, T.D. Rognlien, M.E. Fenstermacher et al., UCRL-JC-147590 and “Hierarchy Tests of Edge Transport Models”, by J. Riemann, M. Borchardt, R. Schneider, X. Bonnin, A. Mutzke, T. Rognlien, M. Umansky, UCRL–JC–151223 are attached. During this research, our German collaborators worked at LLNL for 5 man-months at their expense, and provided local per diem support for our visits (1 man-month) to their institution in Greifswald, Germany.

Survey of stellarator results

The database work combined searches of the published literature from the W7-AS (Germany) and LHD (Japan) stellarator experiments and direct discussions with members of the W7-AS experimental team. Two trips to the W7-AS facilities (as addenda to other trips to Europe for conferences etc.) yielded data on the previous experiments with limiter operation (similar to the initial NCSX operation plans) and new, unpublished data on the first baffled divertor operation of a stellarator. An example from the limiter operation is shown in Fig. 1. An active collaborative relationship now exists between the LLNL stellarator group and the W7-AS experimental group.

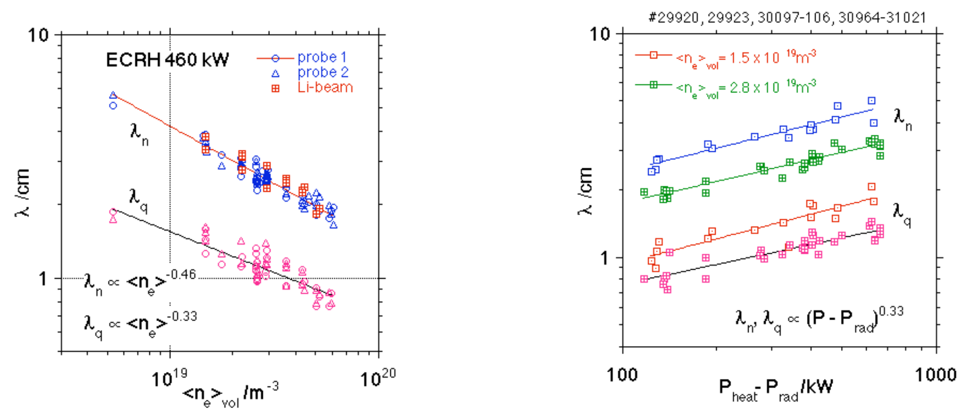

Fig. 1. Density and heat flux scale widths in the W7-AS SOL during limiter-bounded operation vs. volume averaged density and SOL input power (from P. Grigull et al., Proc. 10th Int. Stellarator Conf., Madrid 1995).

Expertise Contributed to NCSX Physics Validation and Conceptual Design Reviews

As a result of the LDRD work in surveying stellarator experimental results, field line tracing with diffusion, and benchmarking the 3D fluid code BORIS against the LLNL 2D fluid code UEDGE, sufficient expertise was developed in stellarator boundary plasma characteristics that we were asked to contribute technical sections to the NCSX Physics Validation Review (PVR) and the Conceptual Design Review (CDR) documents. Analytic modeling of the NCSX boundary plasma was incorporated in the PVR to support the need for long connection length between the last closed flux surface of the core plasma and the material walls (see Ref 1). The conclusion from this work was that the nearly conformal wall, as specified in the NCSX design at the time of the PVR, would not provide sufficient connection length to assure the required boundary plasma characteristics for the experiment. The implications of insufficient connection length on boundary and core plasma performance, specified as part of this work, were: 1) for high core input power, the temperature at the target plate would be excessively high causing large sputtering of the surfaces, leading to impurity contamination of the core plasma, or 2) for acceptable target temperatures, the input power would need to be constrained such that that the upstream temperature near the edge of the core plasma would be low enough to be subject to radiative collapse (see Ref 1).

The diagnostics requirements for the NCSX CDR (see Ref. 4 and Appendix A) were generated from our experience on tokamak boundary plasma experiments and from work on the stellarator boundary plasma experimental results. At the time of the CDR the NCSX project was specifying 6 different phases of operation for the device. Diagnostics essential to the basic success of the experiments in each phase were specified, and additional highly desirable and highly advanced diagnostics were also described (Appendix A). Essential diagnostics included survey cameras with line emission interference filters, residual gas analyzers, filterscopes, surface thermocouples, IRTV cameras, Langmuir probe arrays, reciprocating probes, bolometer arrays, visible spectrometers, and Penning gas-pressure gauges. We emphasized, as part of this work, that the LLNL divertor physics experimental team has experience working with all of these diagnostics in a tokamak environment and that we have constructed many of these systems on other facilities. The full report on the boundary plasma conclusions from the CDR report, "Plasma Boundary Considerations for the National Compact Stellarator Experiment," by P. Mioduszewski, A. Grossman, M. Fenstermacher, A. Koniges, L. Owen, T. Rognlien, M. Umansky, UCRL–JC–151225 is attached.

Conclusions

We have made significant progress in the development of three-dimensional edge-plasma modeling capability in 3D magnetic topologies and performed initial divertor-plasma analyses for the planned National Compact Stellarator Experiment (NCSX). This was the first step toward a significant collaboration with the Princeton Plasma Physics Laboratory, who is building the experimental device. We have used the work from this LDRD project to establish a significant role for LLNL in the design, construction, and physics analysis of boundary plasma diagnostics and other divertor hardware for this device with funding from the DOE Office of Fusion Energy Science (OFES).

The LDRD funds supported adapting stellarator design codes to predict edge plasma conditions in the NCSX stellarator, adding capability to the 3D BORIS edge-plasma transport code, and partnering in the development of a conceptual design for the NCSX stellarator divertor. This project built on the unique, internationally recognized, capabilities of the LLNL MFE program for 2D axisymmetric divertor physics studies combining hardware design, experimental measurements, and numerical simulations for tokamak devices. This project has expanded our competency in simulations of magnetically confined boundary plasmas from two-dimensional to three-dimensional geometry. We now have the capability to perform 3D field-line-tracing with models of cross-field diffusion. The capability also exists to perform 3D fluid simulations of magnetic confinement devices with the BORIS code running at LLNL, including a 3D neutrals model developed by us under this LDRD project. As part of this work, we leveraged our collaboration with the German stellarator program to significantly increase our capability at LLNL.

In support of the NCSX project we have supplied analytic modeling to guide the design and expertise on stellarator experimental results and NCSX diagnostic needs. This new computational capability combined with our recognized expertise in boundary plasma diagnostic hardware and data analysis places us in excellent position to partner with PPPL on the NCSX project. Even in a tight-budget period, we have been awarded an initial grant directly from the NCSX project to contribute to the NCSX boundary plasma design, and we anticipate that this involvement will grow as the project proceeds.

Acknowledgements

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Publications

[1] A. Koniges, A. Grossman, M. Fenstermacher, J. Kisslinger, P. Mioduszewski, T. Rognlien, E. Strumberger, and M. Umansky, “Magnetic Topology of a Candidate NCSX Plasma Boundary Configuration,” UCRL–JC-147845, accepted for publication in Nucl. Fus., 2002.

[2] M.V. Umansky, T.D. Rognlien, M.E. Fenstermacher, M. Borchardt, A. Mutzke, J. Riemann, R. Schneider, L. Owen, “Modeling of Localized Gas Injection in 3D Geometry,” UCRL-JC-147590, accepted for publication in J. Nucl. Mater., 2003.

[3] J. Riemann, M. Borchardt, R. Schneider, X. Bonnin, A. Mutzke, T. Rognlien, M. Umansky, “Hierarchy Tests of Edge Transport Models”, UCRL-JC-151223, accepted for publication in J. Nucl. Mater., 2003.

[4] P. Mioduszewski, A. Grossman, M. Fenstermacher, A. Koniges, L. Owen, T. Rognlien, M. Umansky, “Plasma Boundary Considerations for the National Compact Stellarator Experiment,” s-JC-151225, accepted for publication in J. Nucl. Mater., 2003.

Appendix A – NCSX Diagnostics Description for the CDR Report

In any experimental program, the list of proposed diagnostics contains those that are essential to the operation of the device, those that are highly desirable to obtaining the physics understanding required from the experimental program and more advanced diagnostics that give information on detailed physics issues. This situation is also true for the NCSX project. In addition, the construction and operation of NCSX as a new device is anticipated to occur in phases, each of which having a particular set of machine capabilities and physics objectives. The proposed boundary plasma diagnostics described below are organized first in terms of the phase of NCSX operation to which they would be applied and then categorized as essential, highly desirable or advanced.

Phase 1, Initial Operation Shakedown

The goal of this initial operation phase will be to produce the first plasma in the device. The objective as far as the boundary plasma is concerned will be to document the production of the plasma and to observe as much of the boundary plasma / material wall interaction as possible.

The essential boundary diagnostics in this phase are visible survey cameras. The goal of the cameras will be to document the production of the first plasma by recording images of light emitted from the SOL and to determine the locations of plasma wall interaction both toroidally and poloidally. This provides the basic feedback to initial attempts to position the plasma properly in the vacuum vessel. The survey cameras at this stage should have a tangential view so that as much of the toroidal extent of the plasma as possible can be documented. They should be spatially calibrated so that the precise location of image features that appear during the first plasma discharges can be determined. The camera systems should be outfitted with selectable neutral density attenuation filters so that the image intensity can be kept below saturation. Despite the complicated 3D geometry of NCSX a broad tangential view of the plasma should provide valuable information to guide proper positioning of the initial plasma in the vacuum vessel.

Phase 2 Vacuum and Field Line Mapping

The critical objectives of this phase as far as the plasma boundary is concerned are to verify the vacuum conditions, including an assessment of the initial wall conditions (outgassing, pumping, recycling, impurity generation), to validate the calculations of field line mapping, and to locate regions of excessive plasma wall interaction. There is no plasma operation planned for this phase.

The essential boundary diagnostics in this phase are the visible survey cameras and the vacuum system pressure and residual gas analyzer (RGA) gauges. The goal of the visible cameras is to determine the locations and characteristics of plasma wall interaction both toroidally and poloidally. This phase will be dominated by electron beam mapping of flux surfaces and attempts to predict regions of future plasma wall interaction. It may be possible to use the survey cameras to detect the locations of interaction of the electron beam with the plasma facing surfaces as an indication of the future strikepoints of the plasma on the wall. The pressure and RGA gauges will be used to detect the effect of various wall conditioning procedures on the quality of the vessel vacuum.

Phase 3 Ohmic Operation

The boundary plasma goals during this phase are to assess the effect of plasma wall interaction (both location and source rate) on core plasma performance, to achieve control of the plasma wall contact locations, and to begin the characterization of the SOL plasma. Sufficient information on the parameters of the SOL plasma must be obtained in this phase to guide the design of an optimized second generation of plasma facing components hardware (PFCs) to be installed in the 5th phase of the project.

The additional essential boundary diagnostics for this phase include line integrated visible filterscopes and near surface thermocouples in the target plates. Given success with optimizing the plasma positioning using the visible survey cameras, the target regions of interest should be the tips of the banana cross sections and the inner midplane of the bullet cross section where the limiters are. The boundary diagnostics should be concentrated in these regions.

The purpose of the filterscopes is to determine the source rate of recycled neutrals at critical locations where the neutrals can easily penetrate the core plasma. Therefore an array of filterscopes should view the tip of the banana cross section and another array should view the inner midplane of the bullet cross section. The filterscopes should be equipped with both D_alpha line filters so that the main ion recycling fluxes can be calculated, and also with filters for impurity lines such as CII (514 nm) and CIII (465 nm). Control of the production and transport of recycling neutrals will be the primary goal of the second generation of the PFCs. These will be installed in the tips of the banana cross section during the 5th phase of the project.

A first measurement of target plate heat flux is essential in this phase so that the power balance of 3D simulations can begin to be compared with experiment. These validated computer simulations are needed to guide the design of the 2^{nd} generation PFCs to be installed in Phase 5. This first measurement can be done with an array of thermocouples mounted in the tiles of the banana tip regions. Arrays should be mounted in both banana tips to check the symmetry of the heat flux between the future divertor regions.

Finally, highly desirable boundary diagnostics in this phase include IRTV cameras, Langmuir probe arrays (LPs) in the plasma wall contact region (“target plates”) and a moveable probe of the SOL plasma especially in the banana tip regions. These diagnostics will provide, respectively, the target region heat flux profile with good spatial resolution, the target plate particle flux profile and samples of the spatial profiles of ne, Te and plasma flow velocity in the region where the optimized PFC hardware will be installed. In each case, the measurement contributes the detailed plasma characteristics needed to validate the computer simulations of the boundary plasma in the region where the new PFC hardware hopes to achieve power and particle control in Phase 5.

Phase 4 Auxiliary Heating

The goal of this phase as far as the boundary plasma is concerned is to re-characterize the SOL plasma in the presence of substantial auxiliary heating power. This includes determining the effect of heating power on wall conditions, evaluating the efficacy of wall coating techniques for improving wall conditions with heating power, and examining the effect of biasing parts of the wall on the SOL performance and core confinement.

For this stage all of the essential diagnostics from the previous stages are needed and the three desirable diagnostics from the ohmic phase (IRTV cameras, LP arrays in the targets and a moveable probe) are now essential. The heating conditions during this phase will be close to those used with the 2nd generation PFCs (although the heating power will be somewhat lower) so complete measurements of the plasma conditions in the banana tips are required to complete the design of the new hardware.

The pressure of the neutral gas in both the banana tips and near the midplane of the bullet cross section should also be measured during this phase. This will require at least two ASDEX type pressure gauges with time response short compared with the plasma transitions induced by application of the auxiliary heating power. It would be desirable to have a third ASDEX gauge in the opposing banana tip region to check for up/down asymmetries.

To examine the effect of wall coating techniques on the boundary plasma the filterscopes and visible cameras should be upgraded to include imaging of some of the wall coating elements. Filters for visible lines of lithium (neutral and singly ionized) and boron should be available. If boronization is used then visible lines of helium should also be monitored since substantial helium is trapped in the boronization process and multiple discharges can be required after boronization to bring the helium concentration down to normal levels.

Finally, highly desirable boundary diagnostics in this phase include bolometer measurements, a visible survey spectrometer and a reciprocating probe of the SOL plasma especially in the banana tip regions. These diagnostics will provide, respectively, the total radiated power, the relative contributions to the radiated power from separate constituents of the SOL plasma (vis. deuterium neutrals, carbon species and other impurities), and detailed spatial profiles of ne, Te and plasma flow velocity in the region where the optimized PFC hardware will be installed. In each case, the measurement contributes the detailed plasma characteristics needed to validate the computer simulations of the boundary plasma in the region where the new PFC hardware hopes to achieve power and particle control in Phase 5.

Phase 5 Confinement and Beta Push

At some point during this phase a new set of plasma facing components will be installed. The goal of the new PFCs will be to handle the high heat loads that result from substantial auxiliary heating of the core plasma and to control (minimize) the influx of recycling neutrals and sputtered impurities from the plasma targets back to the core plasma. This hardware will function much like a poloidal divertor in tokamaks including target surfaces optimized to handle high heat flux, baffle structures to restrict escape of neutrals from the divertor region back to the core, and possibly active pumping of neutrals to control the SOL and core plasma density. In addition, the potential for pumping of helium exhaust from the core will be examined in this phase by injecting trace levels of helium.

The list of essential diagnostics for this phase includes the divertor region bolometer arrays, the visible spectrometer measurement of the radiating constituents in the divertor plasma and the divertor reciprocating probe. The new diagnostics that are essential in this phase are: 1) Penning gauges in the divertor plenum and midplane, and 2) simultaneous measurements of D_alpha with either D_beta or D_gamma emission from the divertor region. The Penning gauges will be used to determine the fractions of various neutral gasses in the pumping plenum compared with the SOL. The ratio of these concentrations is the divertor enrichment. The enrichment of helium in the divertor is the critical parameter for assessment of future helium exhaust schemes in compact stellarators.

Simultaneous measurement of multiple deuterium emission lines in the divertor provides a technique to quantify the level of detachment of the divertor plasma from the target surfaces. The ratio of the emission intensities from the different lines changes substantially for a detached plasma (high density and ~ 1eV temperature) compared with an attached divertor plasma (moderate density and > 10 eV temperature). Experiments will be done during this phase to determine the available parameter regime for detached divertor operation of NCSX. For future high power compact stellarators, as in tokamaks, it is likely that the plasma will need to be detached from the target surfaces to reduce the target heat fluxes to acceptable levels.

During the detachment experiments in this phase it will be highly desirable to have increased spatial resolution in the divertor region of radiated power measurements, constituent line emission measurements, target probe data. These upgrades will resolve the strong gradients between the hot upstream SOL plasma and the detached plasma near the target surfaces. These requirements can be achieved simply by increasing the number of lines of sight of the existing bolometer arrays, filterscope arrays and target mounted Langmuir probe arrays in the divertor region.

It will also be highly desirable to have volumetric measurements of electron density and temperature and measurements of the UV emission from various constituents in the divertor. Measurements of ne and Te within the plasma above the target surfaces can be done, albeit in a perturbative way, with the reciprocating probe passing through the plasma above the divertor target. However, it would be desirable to have a Thomson scattering system in one of the divertors with measurements at a number of locations vertically off of the divertor target to provide non-perturbative measurements of ne and Te in the attached and detached divertor plasma. The measurement of UV emission in the divertor is needed because these lines will contain the majority of the radiated power in the divertor during detached operation. A UV SPRED spectrometer with a wavelength range of 10 – 160 nm would provide the required data. It would also be desirable to install UV survey cameras with views of the divertor to provide the spatial profile of the emission lines containing the largest fraction of the divertor radiated power. Previous experience with tokamaks indicates the radiated power will likely be dominated by CIV

(155nm) and L_alpha (121nm) emission, both of which can be imaged with MgF2 optical components, UV-to-visible conversion phosphors and standard visible cameras.

Phase 6 Long Pulse

The goal of this phase of NCSX operation is to extend the highest performance of the device to many energy confinement times. As far as the boundary plasma this means that the goal is to reach steady conditions near the optimum for core performance and not produce any increasing impurity sources or damage of target surfaces during long high power discharges. A 3rd generation of the PFC hardware may be installed at some point during this phase. Optimization of this final PFC design will come from the measurements and validated code modeling done during Phases 4 and 5.

The focus of the boundary diagnostics in this phase will be on the conditions of the target material as the discharge length increases. Since toroidal non-uniformity will be important in this phase, the number of IRTV systems viewing the targets should be increased to cover all the targets. Each target should also be equipped with Langmuir probes.

With steady long pulse conditions in this phase it may be possible to do detailed investigations of SOL and divertor flows. The advanced diagnostics for measuring plasma flow in the boundary either use charge exchange emission from diagnostic neutral beams or Laser Induced Fluorescence (LIF) of impurity ions in the boundary plasma. Both techniques are being developed for tokamaks at the present time. The optimum technique to use for NCSX will be determined after they have been tried on present devices.